NON-PUBLIC?: N
ACCESSION #: 8910040398
LICENSEE EVENT REPORT (LER)

FACILITY NAME: PLANT HATCH, UNIT 2 PAGE: 1 OF 6

DOCKET NUMBER: 05000366

TITLE: FEEDWATER CONTROLLER FAILURE CAUSES REACTOR SCRAM ON LOW WATER
LEVEL
EVENT DATE: 09/03/89 LER #: 89-005-00 REPORT DATE: 09/27/89

OPERATING MODE: 1 POWER LEVEL: 070

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE:X SYSTEM: JB COMPONENT: LIK MANUFACTURER: G084
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 9/3/89 at approximately 2239 CDT, Unit 2 was in the Run mode at an approximate power level of 1695 CMWT (approximately 70% of rated thermal power). At that time, licensed operations personnel were changing reactor vessel water level control from single element to three element control following completion of procedure 57SV-SUV-004-2S, "Excess Flow Check Valve Operability," for the main steam line flow instruments' excess flow check valves. When the master controller was placed in Automatic following the change from single element to three element control, the controller's output signal suddenly went to zero. Both Reactor Feed Pumps decreased feedwater flow to the reactor vessel in response to the controller's zero output signal. Reactor vessel water level decreased and the reactor scrammed on low water level.

The root cause of this event is component failure. The Self Synchronized

Control Unit, the main operating unit of the master controller, failed when the master controller was placed in Automatic. The failure of the Self Synchronized Control Unit caused the master controller's output signal to go to zero.

Corrective actions for this event included replacing the Self Synchronized Control Unit.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code XX).

SUMMARY OF EVENT

On 9/3/89 at approximately 2239 CDT, Unit 2 was in the Run mode at an approximate power level of 1695 CMWT (approximately 70% of rated thermal power). At that time, licensed operations personnel were changing reactor vessel water level control from single element to three element control following completion of procedure 57SV-SUV-004-2S, "Excess Flow Check Valve Operability," for the main steam line flow instruments' excess flow check valves (EIIS Code BD). When the feedwater system's master controller (EIIS Code JB) was placed in Automatic following the change from single element to three element control, the controller's output signal suddenly went to zero. Both Reactor Feed Pumps (RFPs, EIIS Code SJ) decreased feedwater flow to the reactor vessel in response to the controller's zero output signal. Reactor vessel water level decreased and the reactor scrammed on low water level. The cause of this event is component failure. The Self Synchronized Control Unit (EIIS Code JB), the main operating unit of the master controller, failed. The unit was replaced.

DESCRIPTION OF EVENT

On 9/3/89, non-licensed Instrument and Control (I&C) technicians were performing procedure 57SV-SUV-004-2S, "Excess Flow Check Valve Operability," for the main steam line flow instruments' excess flow check valves (EFCVs). The procedure requires the feedwater system's master controller to be placed in single element (reactor water level) control before testing these EFCVs. The master controller's mode switch was in its normal position of three element (feedwater flow, main steam line flow, reactor water level) control prior to testing. The master controller had to be placed in single element control because one of the signals (main steam line flow) necessary for proper three element control would not be available during EFCV testing.

At approximately 2227 CDT, the I&C technicians successfully completed testing of the main steam line flow instruments' EFCVs and, per procedure 57SV-SUV-004-2S, notified the on-shift Plant Operator that the master controller mode switch could be positioned as desired. The operator chose to position the switch to its normal position of three element control. Per procedure 34SO-N21-007-2S, "Condensate and

Feedwater System," the Plant Operator placed the master controller in manual, placed the mode switch in the three element position, nulled the deviation meter (i.e., matched the input and demand, or output, signals), and placed the master controller in Automatic.

At approximately 2239 CDT, as the Plant Operator was placing the master controller in Automatic, the output from the master controller unexpectedly went to zero. This change in the master controller's output caused the RFPs to decrease feedwater flow to the reactor vessel based on a decreased demand for feedwater from the master controller. Reactor water level decreased as feedwater flow decreased. The reactor water level low scram setpoint of approximately 12 inches above instrument zero was reached about eight seconds after the controller failed, resulting ,in an automatic reactor scram.

Reactor water level decreased to approximately minus 35 inches as measured from instrument zero (129 inches above the top of the active fuel) before the licensed Assistant Plant Operator was able to gain manual control of the "A" RFP and restore level. The High Pressure Coolant Injection (HPCI, EIIS Code BJ) and Reactor Core Isolation Cooling (RCIC, EIIS Code BN) systems received a start signal on low low water level and started, but did not inject into the reactor vessel. These system responses were consistent with design since reactor water level recovered to a point above the initiation setpoint (minus 35 inches) prior to the injection valves receiving an open signal. HPCI and RCIC logic is such that the low reactor water level initiation signal goes to the Steam Inlet Valve as well as the Injection Valve. However, the Injection Valve also needs signals (i.e., valve, off seat) from the Steam Inlet Valve'and Turbine Stop Valve (Turbine Trip and Throttle Valve on RCIC) before it will begin to open. Reactor water level had recovered to a point above the initiation setpoint before the Injection Valve received the other necessary signals. The low reactor

water level signal to the Injection Valve is not a seal-in signal.
Therefore, this signal cleared before the Injection Valve could receive its other opening signals and begin to open.

In addition to the HPCI and RCIC systems auto starting, both the "A" and
the "B" Recirculation Pumps (EIIS Code AD) tripped, the Reactor Building normal ventilation system (EIIS Code VA) inboard dampers isolated, the "A" train of the Standby Gas Treatment (SBGT, EIIS Code BH) system auto started, Group 2 Primary Containment Isolation System (PCIS, EIIS Code JM) valves isolated, and one half of the Group 5 PCIS valves isolated.

The "B" train of the SBGT system did not auto start, the normal ventilation system outboard isolation dampers did not isolate, and one half of the Group 5 PCIS valves did not isolate. These actuations did not occur because the level sensors which input into the "B" SBGT system logic (which, in turn, initiates the isolation of the normal ventilation

outboard isolation dampers) and the Group 5 PCIS logic did not sense the minus 35 inches reactor water level. The setpoints for the instruments that would cause these actuations were not reached due to setpoint tolerance and rapid recovery of the reactor water level.

Field checks of level transmitters 2B21-NO81A through D and Analog Transmitter and Trip System (ATTS, EIIS Code JF) trip units 2B21-N682A through D, which provide signals to the logic systems of interest, revealed the above systems functioned correctly. Each trip unit's setpoint is established based on a setpoint methodology which provides a margin between the actual setpoint and the limit in the Technical Specifications. This margin consists of allowances for expected transmitter and trip unit drift and provides a 'leave alone band' referred to as the instrument's tolerance (an instrument found within this band during calibration does not need to have the setpoint adjusted).

The field checks confirmed the transmitters, except for 2B21-NO81A, and the trip units' trip setpoints were within their tolerances. The trans mitter 2B21-NO81 A was only slightly out of its tolerance band (less than 1/4% of full scale). However, 2B21-NO81A was well within the overall margin allowed for drift and it was associated with the logic that did actuate. The transmitter's allowable tolerance of + 0.5 inches and the reactor vessel water level decreasing briefly to minus 35 inches before increasing indicated that the level transmitted to the trip units which did not actuate could have been as high as minus 34.5 inches. This would not have caused the trip units to actuate. Because the level decreased to a point very close to the trip point for several systems and considering the allowable tolerance of the transmitters, the system responses were as expected.

Recovery from the scram was normal. The main turbine (EIIS Code TA) continued to deplete main steam for several seconds following the scram until it was tripped manually. The "A" RFP turbine also depleted steam until the Main Steamline Isolation Valves (MSIVs, EIIS Code JM) were closed to control the vessel cooldown rate. The HPCI and RCIC systems were used for pressure control after the MSIVs were closed. Neither the Bypass Valves (EIIS Code SO) nor the Safety Relief Valves were needed to control pressure since pressure never increased above the normal operating value of approximately 980 psig.

The "A" RFP was used in manual control to restore and then maintain reactor vessel water level until the MSIVs were closed. HPCI and RCIC were used to control water level thereafter.

CAUSE OF THE EVENT

The root cause of this event is component failure. The Self Synchronized Control Unit, the main operating unit of the feedwater system's master controller, failed when the master controller was placed in Automatic. The sudden failure may have been the result of relays internal to the Control Unit sticking when the master controller was transferred from Manual to Automatic. If these relays stuck in position and did not engage when the master controller was placed in Automatic, an open circuit would exist in the controller and its output would be expected to drop suddenly to zero.

Following the event and during initial troubleshooting activities, it was found the master controller had an output signal when in Manual, but not in Automatic. This is indicative of stuck relays in the Self Synchronized Control Unit. The entire feedwater master controller was removed from the Main Control Room and bench tested in the I&C shop. The controller then appeared to function correctly, probably because in removing and transporting the controller the relays were freed. Upon re-installation in the Main Control Room, the controller's output signal slowly decreased to zero and would not respond to any demand changes. This indicates there were other problems internal to the unit. It was decided to replace the entire Self Synchronized Control Unit and send the defective unit to the manufacturer for further analysis and repair, if possible.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required per 10 CFR 50.73(a)(2)(iv) because an unplanned actuation of the Reactor Protection System (RPS, EIIS Code JC) and Engineered Safety Features (ESFs) occurred. Specifically, the RPS was initiated automatically on low reactor water level. The ESFs which activated during this event were the Primary Containment Isolation System valve Group 2 and Group 5 (partial), the High Pressure Coolant Injection system (although it did not inject), and the "A" train of the Standby Gas Treatment system.

The RPS provides timely protection, against the onset and consequences of conditions that could threaten the integrity of the fuel barriers and the nuclear system process barrier. A reactor scram initiated by a low water level condition protects the fuel by reducing the fission heat generation within the core. In this event, the decrease in vessel level was a direct result of the failure of the feedwater system master controller. The RPS functioned per design. Reactor water level was restored quickly by using the "A" Reactor Feed Pump, and maintained by

the High Pressure Coolant Injection system, and the Reactor Core Isolation Cooling system. At no time was water level less than 129.5 inches above the top of the active fuel. All systems functioned as designed to restore water level to its normal level. Based on this information, it is concluded that this event had no adverse impact on nuclear safety. Additionally, the above analysis is applicable to all power levels.

CORRECTIVE ACTIONS

The Self Synchronized Control Unit was replaced with a new unit. Additionally, the slidewire and manual and auto pushbuttons on the Manual/Automatic Transfer Station were replaced. Several capacitors in the Control Amplifier were also replaced. The entire master controller was calibrated per procedure 57CP-CAL-226-2N, "Feedwater Control Calibration for Master and Bypass Valve Loops," and returned to service.

ADDITIONAL INFORMATION

1. Failed Component Identification
MPL: 2C32-K636, Self Synchronized Control Unit
Manufacturer: General Electric Company Root Cause Code: X
Model Number: 547-12 Component Code: LIK

Type: N/A Manufacturer Code:G084
EIIS Code: JB
Reportable to NPRDS: Yes

2. Previous Similar Events

There were four previous similar events in which loss of feedwater resulted in low reactor vessel water level and a reactor scram. These events were reported in LER 50-321/1988-013 dated 10/3/88, LER 50-366/1988-008 dated 4/20/88, LER 50-366/1988-017 dated 6/27/88, and LER 50-366/1988-020 dated 9/6/88. The corrective actions for these four events would not have prevented this event because the root causes were different. This event resulted from a failed Self Synchronized Control Unit whereas two of the previous events resulted from deficient procedures, one resulted from a blown fuse in the feedwater control circuit, and one resulted from bad or loose connections in the control circuits for the individual feedpumps.

ATTACHMENT 1 TO 8910040398 PAGE 1 OF 2

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September 27, 1989

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
FEEDWATER CONTROLLER FAILURE CAUSES
REACTOR SCRAM ON LOW WATER LEVEL

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv),
Georgia Power Company is submitting the enclosed Licensee Event Report
(LER) concerning the unanticipated actuation of some Engineered Safety
Features (ESFs). This event occurred at Plant Hatch - Unit 2.

Sincerely,

W. G. Hairston, III

JJP/ct

Enclosure: LER 50-366/1989-005

c: (See next page.)

ATTACHMENT 1 TO 8910040398 PAGE 2 OF 2

GEORGIA POWER

U.S. Nuclear Regulatory Commission
September 27, 1989
Page Two

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GO-NORMS

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